Entergy

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Anthony J. Vitale
Site Vice President

NL-18-083

November 19, 2018

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop O-P1-17
Washington, D.C. 20555-0001

Subject: Licensee Event Report # 2018-003-00 "Manual Reactor Trip Due To A Steam Leak On A High Pressure Feedwater Heater"
Indian Point Unit No. 3
Docket No. 50-286
DPR-64

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc, hereby provides Licensee Event Report (LER) 2018-003-00. The attached LER identifies an event where the reactor was manually trip due to a steam leak on a high pressure feedwater heater, which is reportable under 10 CFR 50.73(a)(2)(iv)(A). This event was recorded in the Entergy Corrective Action Program as Condition Report CR-IP3-2018-02773.

There are no commitments made or revised in this letter. Should you have any questions regarding this matter, please contact Mr. Robert Walpole, Manager, Regulatory Assurance, Indian Point Energy Center at (914) 254-6710.

Sincerely,

AJV/trj McCarthy for
V. Vitale

cc: Mr. David Lew, Regional Administrator, NRC Region I
NRC Resident Inspector's Office
Ms. Bridget Frymire, New York State Public Service Commission

IE22
NRR

NRC FORM 366 (04-2018) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 03/31/2020

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

(See NUREG-1022, R.3 for instruction and guidance for completing this form http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1022/r3/)

1. Facility Name	2. Docket Number	3. Page
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4. Title
MANUAL REACTOR TRIP DUE TO A STEAM LEAK ON A HIGH PRESSURE FEEDWATER HEATER

5. Event Date			6. LER Number			7. Report Date			8. Other Facilities Involved	
Month	Day	Year	Year	Sequential Number	Rev No.	Month	Day	Year	Facility Name	Docket Number 05000
9	18	2018	2018	- 003	- 00	11	19	2018	Facility Name	Docket Number 05000

9. Operating Mode: 1

10. Power Level: 100

11. This Report is Submitted Pursuant to the Requirements of 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(C)	☐ 73.77(a)(1)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(D)	☐ 73.77(a)(2)(i)
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(vii)	☐ 73.77(a)(2)(ii)
	☐ 50.73(a)(2)(i)(C)	☐ Other (Specify in Abstract below or in NRC Form 366A)	

12. Licensee Contact for this LER

Licensee Contact	Telephone Number (Include Area Code)
Nelson Azevedo, Supervisor, Code Program Engineering	914-254-6775

13. Complete One Line for each Component Failure Described in this Report

Cause	System	Component	Manufacturer	Reportable To ICES	Cause	System	Component	Manufacturer	Reportable To ICES
B	SJ	PSF		Y					

14. Supplemental Report Expected	15. Expected Submission Date	Month	Day	Year
☐ Yes (If yes, complete 15. Expected Submission Date) ☒ No				

ABSTRACT

On September 18, 2018, with the reactor at 100 percent power, at approximately 0525 hours Operators manually tripped the reactor and shut the Main Steam Isolation Valves due to a steam leak on the 6 inch elbow located upstream of the 36C Feedwater Heater. The steam leak was due to a failure of the 6 inch elbow. The cause of the failure was flow accelerated corrosion which lead to pipe wall thinning and subsequent pipe failure. Ultrasonic testing of the elbow showed flow accelerated corrosion thinning in the failed elbow and also in the adjacent upstream elbow.

This event was reported under 10 CFR 50.72(b)(2)(iv)(B) for any event or condition that results in actuation of the reactor protection system when the reactor is critical. This event was also reportable under 10 CFR 50.72(b)(3)(iv)(A) for any event or condition that results in valid actuation of any of the systems listed in paragraph 10 CFR 50.72(b)(3)(iv)(B). This included actuation of the Auxiliary Feedwater System as expected following manual reactor trip. Following the reactor trip, the plant was stabilized in hot standby with decay heat being removed from the steam generators via the Auxiliary Feedwater System and atmospheric steam dumps.

This event had no effect on the public health and safety.

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		YEAR	SEQUENTIAL NUMBER	REV NO.
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NARRATIVE

On September 18, 2018, with the reactor at 100 percent power, at approximately 0525 hours Operators manually tripped the reactor and shut the Main Steam Isolation Valves due to a steam leak on the 6 inch elbow located upstream of the 36C Feedwater Heater. The steam leak was due to a failure of the 6 inch elbow. The cause of the failure was flow accelerated corrosion which lead to pipe wall thinning and subsequent pipe failure. Ultrasonic testing of the elbow showed flow accelerated corrosion thinning in the failed elbow and also in the adjacent upstream elbow. The piping containing the failure is in the drain path from the Moisture Separator Reheater Drain Tank to the 36A, B, and C Feedwater Heaters. This component is identified as RHD-02, 15A-06E. The failure occurred on the extrados of the elbow and was approximately 1.75 inches in length.

The flow accelerated corrosion program as described in Entergy procedure, EN-DC-315, "Flow Accelerated Corrosion Program" is designed to predict, detect, monitor and minimize degradation in single and two-phase flow piping (safety and non-safety related systems) to prevent failures while enhancing plant safety and reliability. Flow accelerated corrosion is monitored through the CHECKWORKS™ Steam/Feedwater Application database. This database was developed by the Electric Power Research Institute (EPRI) as a formal software plan and is used by the nuclear and fossil industries.

The CHECKWORKS™ Steam/Feedwater Application model is a predictive methodology tool used to predict the rate of wall thinning due to flow accelerated corrosion within piping and fittings under exact operating conditions. The predicted wear rate and remaining service life are based on factors such as component geometry, material, and operating conditions. The Indian Point flow accelerated corrosion program was developed consistent with the industry recommended program as outlined in EPRI procedure NSAC-202L.

The flow accelerated corrosion program was ineffective in detecting and correcting the flow accelerated corrosion on the elbow of 36C Feedwater Heater Branch prior to the failure because the flow accelerated corrosion engineers did not use system replacement history to identify differing wear rates resulting from differing operating conditions. In addition, there were weaknesses in the setup of the CHECKWORKS™ model for the affected system. Specifically, all six Re-Heater Drain branches were modeled in a single run and the internal fluid frictional losses in some branches of the system were being subjected to higher flow velocities than other branches. As a result, higher wear rates in the 36C Feedwater Heater piping were being masked by the lower wear rates in the 36A and 36B Feedwater Heater piping. These conditions will be corrected by revising the CHECWORKS models for multi branch systems and by improving the fleet procedure with additional actions.

This event was reported under 10 CFR 50.72(b)(2)(iv)(B) for any event or condition that results in actuation of the reactor protection system when the reactor is critical. This event was also reportable under 10 CFR 50.72(b)(3)(iv)(A) for any event or condition that results in valid actuation of any of the systems listed in paragraph 10 CFR 50.72(b)(3)(iv)(B).

This event was recorded in the Indian Point Energy Center Corrective Action Program as CR-IP3-2018-02773.

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CAUSE OF EVENT

The direct cause of the failure of the elbow was flow accelerated corrosion. The elbow wall thinning occurred because the flow accelerated corrosion continued until the remaining wall thickness was insufficient to withstand the internal system pressure.

Root Cause
The qualified flow accelerated corrosion program engineers did not use the system replacement history to identify the fact that the Re-Heater Drain piping to the 36C feedwater heater was more susceptible to the flow accelerated corrosion then the Re-Heater Drain piping to the 36A and 36B feedwater heaters. Plant operating experience showed failures had occurred in 2007 on the 36C feedwater heater branch. As a result of this, additional inspections should have been performed during subsequent refueling outages on the 36C train, compared to the 36A and 36B trains to monitor for subsequent wall thinning. This action could have identified the thinning prior to failure.

Contributing Causes:

- The Entergy flow accelerated corrosion program was ineffective in detecting and correcting flow accelerated corrosion in the elbow prior to failure because of weaknesses in the setup of the CHECKWORKS™ model. Specifically, all six Re-Heater Drain branches were modeled in a single run and the internal fluid frictional losses in some branches of the system were resulting in higher flow velocities than other branches. As a result of this, higher wear rates in the 36C Feedwater Heater piping were being masked by the lower wear rates in the 36A and 36B Feedwater Heaters piping.

- Inadequate procedural guidance in Entergy procedure EN-DC-315, "Flow Accelerated Corrosion Program" for scope expansion. At the time of the 2007 failure, EN-DC-315 Revision 0) Paragraph 5.12[2](a) required scope expansion to include, "components within two diameters downstream" to be inspected but does not include the next downstream fitting which would have identified thinning on the failed elbow.

CORRECTIVE ACTIONS

The following corrective actions have been or will be performed under the Entergy Corrective Action program to address the causes of this event.

- Revise the CHECKWORKS model to split the six Re-Heater Drain branches into three separate runs, one run per heater. Apply the best estimate thermo-hydraulic conditions to each of the three runs.
- Revise fleet procedure EN-DC-315 to provide more guidance for scope expansion to ensure the extent of the worn area is known.
- Revise fleet procedure EN-DC-315 to give the flow accelerated corrosion engineer a visual of the system replacement history in order to see overall system impact and determine appropriate inspection scope.

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EVENT ANALYSIS

Due to a steam leak on the reheater drain line to the 36C Feedwater Heater, operators initiated a manual trip of the reactor, verified the reactor trip, and closed all Main Steam Isolation Valves. The plant was stabilized in Mode 3 with the steam leak isolated.

This event was reported under 10 CFR 50.72(b)(2)(iv)(B) for any event or condition that results in actuation of the reactor protection system when the reactor is critical. This event was also reportable under 10 CFR 50.72(b)(3)(iv)(A) for any event or condition that results in valid actuation of any of the systems listed in paragraph (b)(3)(iv)(B). This included actuation of the Auxiliary Feedwater System as expected following manual reactor trip. Following the reactor trip, the plant was stabilized in hot standby with decay heat being removed from the steam generators via the Auxiliary Feedwater System and atmospheric steam dumps.

PAST SIMILAR EVENT

A review was performed of the past five year for Indian Point Unit 2 and Unit 3 Licensee Event Reports for flow accelerated corrosion flaws. There were no past similar events. Plant operating experience showed that a pin hole leak occurred in 2007 on the 36C Feedwater Heater Branch but no failures similar to the one described here. However, this leak did not result in a reactor trip.

SAFETY SIGNIFICANCE

This event has no effect on the health and safety of the public. There were no actual safety consequences for the event because it was an uncomplicated manual reactor trip. The required primary safety systems performed as designed.

For the event, all control rods inserted as required upon initiation of the reactor trip. The reactor coolant system remained below the setpoint for pressurizer power operated relief valve and code safety valve operation, and above the setpoint for automatic Safety Injection actuation. Following the reactor trip, the plant was stabilized in hot standby with decay heat being removed from the steam generators via the Auxiliary Feedwater System and atmospheric steam dumps.